U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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## TABLE OF CONTENTS

Table of Contents

## EXECUTIVE SUMMARY

This inspection involved a review of Peach Bottom's implementation of 10 CFR 50.65, the maintenance rule. The report covers a one week on site inspection by regional and NRR inspectors during the week of August 5, 1996.

MAINTENANCE

Overall, the team judged the Peach Bottom maintenance rule program to be weak. Monitoring the effectiveness of maintenance under the maintenance rule did not appear to be fully integrated into the existing maintenance program at Peach Bottom, which has previously been assessed to be excellent. For example, system managers frequently had thorough knowledge of their systems, performed good general monitoring of their systems, and had proposed system changes within other maintenance improvement systems, but the performance criteria under the maintenance rule were weak and often misunderstood by the system managers.

There was a general problem of not documenting the methods, processes and results of maintenance rule implementation. Also, there was a lack of documented justification when not following the NUMARC 93-01 guidance.

A number of performance criteria were found to be ineffective for monitoring performance in accordance with the maintenance rule. This was identified as an apparent violation of the rule. PECO Nuclear did not account for system demands and successful operation as part of reliability criteria, did not have availability criteria for some systems, did not have availability or reliability criteria for the control room emergency ventilation system, did not use the appropriate criteria of unplanned capability loss factor for some systems, did not have criteria for all maintenance rule functions for some systems, and did not monitor MSRVs at the appropriate system level. All of these approaches conflict with NUMARC 93-01, which is the method used by PECO Nuclear for demonstrating the effectiveness of preventive maintenance. Given the problems with establishing and using performance criteria to monitor performance, the team could not conclude that PECO Nuclear was effectively monitoring and trending structures, systems or components (SSC) performance.

The team concurred with all SSCs included within the scope of the rule. Which components of the alternate ac power source are within the scope of the rule was under facility review and is an unresolved item.

The risk determination process for SSCs, which included the expert panel deliberations, was weak. The bases provided to the team for several expert panel decisions on SSC risk significance did not provide supportable bases to deviate from the NUMARC guidelines and generally accepted probabilistic analysis methods, and resulted in an unresolved item. Minimal documentation for the basis for the expert panel's decisions was available. Further, the team had concerns with the use of generic failure probability data and the undocumented basis for the use of the top 85% (versus 90%) of the cut sets contributing to core damage frequency.

Plans and progress for performing periodic evaluations were weak. The preliminary evaluation for Unit 3 lacked details in some important areas such as evaluating the effectiveness of balancing reliability and unavailability, adjustments made to preventive maintenance for SSCs and a summary of SSCs moved from (a)(2) to (a)(1) or vice versa. The approach, if continued to be taken by PECO Nuclear, to balance reliability and unavailability would not appear to provide meaningful estimates of reliability for use in balancing and would not meet the intent of the rule.

Licensed operators and outage planners demonstrated a general understanding of their duties and responsibilities for implementing the maintenance rule, but exhibited some uncertainty about the systems covered by the rule. A better knowledge of which systems were covered by the rule would allow operators and planners to more effectively carry out their maintenance rule responsibilities, including safety assessments prior to removing equipment from service.

A program existed to utilize industry operating experience, and there were cases where industry operating experience had been used in plant modifications. Nonetheless, there was no indication that industry experience had been used when setting goals and performance criteria.

Engineering staff overall knowledge of systems was excellent. Knowledge and use of maintenance rule performance criteria to monitor performance was weak in some cases.

The material condition of the systems examined was generally excellent.

Corrective actions taken to solve maintenance problems were found to be appropriate.

In general, self-assessments were comprehensive and provided meaningful feedback to management. While weaknesses were identified in the self-assessments, some of the problems noted by the team were not identified in the self-assessments. Program improvements were noted in several areas as a result of the self-assessments. Results of the June 1996 assessment were still being acted upon.

## DETAILS

## II. MAINTENANCE

### M1 Conduct of Maintenance (62706)

The primary focus of the inspection was to verify that the PECO Nuclear had implemented a maintenance monitoring program, which satisfied the requirements of the maintenance rule (10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.") The inspection was performed by a team of seven inspectors, which included regional and headquarters inspectors. Additional assistance and support was provided by three members of the Quality Assurance and Maintenance Branch, NRR, and one member of the Probabilistic Safety Assessment Branch, NRR.

### M1.1 SSCs Included Within the Scope of the Rule

a. Inspection Scope

The team reviewed the scoping documentation to determine if the appropriate SSCs were included within their maintenance rule program in accordance with 10 CFR 50.65(b). The team used Inspection Procedure 62706, NUMARC 93-01 and Reg. Guide 1.160 as references during the inspection.

b. Observations and Findings

PECO Nuclear used th PIMS (Plant Information Management System), a computerized data base of all plant systems, to identify the SSCs to be initially considered for placement under the maintenance rule. A total of 326 SSCs was considered in the scoping phase. Of these, 133 SSCs were placed within the scope of the rule with 42 SSCs identified as risk significant.

The 326 SSCs were listed as an attachment to Procedure AG-CG-28.1, Rev. 1, "Maintenance Rule Implementation Program." This listing identified the system number and description of each SSC and assigned the maintenance rule function(s) of each system. In addition, yes or no answers were supplied to each of the following questions that were asked of all SSCs:

Safety related?
Mitigates accidents or transients?
Used in Emergency Operating Procedures (EOPs)?
Could failure prevent a safety related SSC from functioning?
Could failure cause a scram or safety system actuation?
Is the SSC within scope of the maintenance rule?
Is the SSC risk significant?
Is the SSC normally in standby?

The team reviewed appropriate documentation associated with all of the SSCs. This documentation, in part, detailed the justification of the answers to the above listed questions for each SSC. The team determined that PECO Nuclear had correctly identified the plant systems that were required to be scoped within the maintenance rule. However, the documentation detailing the technical basis of some scoping decisions was not well written. PECO Nuclear had prepared an action request (AR 1039453) prior to the NRC inspection that had identified the documentation issue and had planned to reconstitute and consolidate the scoping bases.

The team observed the ongoing activities concerning the scoping of the alternate ac power source, which may lead to its being placed under the rule. The team found that monitoring of availability of this ac power source was satisfactory using existing programs. However, there is a need to define the boundary and components of the alternate ac power source that are within the scope of the rule.

c. Conclusions

The team concluded that PECO Nuclear had correctly identified the SSCs that were required to be within the scope of the rule. There was a need to identify the boundary and components of the alternate ac power source that fall under the rule. The identification of the alternate ac power source boundary and components within the scope of the maintenance rule is an unresolved item. Also, documentation of the technical basis for some scoping decisions was not clearly written. (URI 50-277;278/96-07-01).

M1.2 Safety (Risk) Determination, Risk Ranking, and Expert Panel

a. Inspection Scope

Paragraph (a)(1) of the rule requires that goals be commensurate with safety. Additionally, implementation of the rule using the guidance contained in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires that safety be taken into account when setting performance criteria and monitoring under Paragraph (a)(2) of the rule. This safety consideration would be used to determine if the SSC should be monitored at the system, train or plant level. The team reviewed the methods and calculations that PECO Nuclear had established for making these safety determinations. The team also reviewed the safety determinations that were made for the specific SSCs.

b. Observations and Findings

b.1 Expert Panel

NUMARC 93-01 recommends the use of an expert panel to establish risk significance of SSCs by combining PRA insights with operations and maintenance experience, and to compensate for the limitations of PRA modeling and importance measures. PECO Nuclear used an expert panel process in conjunction with a PRA ranking methodology to determine the risk significance of SSCs within the scope of the rule. The team reviewed the expert panel process and the information available that documented the decisions made by the panel.

One nonrisk-significant and 18 risk-significant SSCs, as evaluated by the PRA, were changed by the expert panel. There were no balance of plant (BCP) SSCs remaining in the risk-significance category following the expert panel determinations. Nonrisk-significant SSCs will be monitored at the plant level versus the system or train level and will not need to be considered during the periodic evaluation of balancing reliability with unavailability. Overall, when considering risk significance by PRA and other methods, the expert panel changed eight non-risk SSCs to risk-significant and 29 risk-significant SSCs to non-risk significant.

Two of the 18 risk significant systems changed by the panel were condensate and feedwater. The feedwater system met NUMARC risk-significance criteria for all three importance measures: (1) risk reduction worth, (2) risk achievement worth, and (3) core damage frequency contribution (RRW, RAW, and CDF cutset contribution), whereas the condensate system met the criteria for RAW. Despite this, the panel's determination was based on the fact that these systems had multiple trains and that other sources of vessel makeup existed (e.g., LPCI, HPSW). However, the PRA modeled feedwater and condensate as single "black box" inputs, which accounted for the reliability of all trains. The PRA also accounted for the other sources of vessel makeup; therefore the appropriate amount of analytical credit for these alternate sources should have already been realized. The team found the expert panel's documented reasons for classifying the feedwater and condensate systems as non-risk significant were inappropriate in that the documentation did not provide a supportable reason for deviating from the NUMARC guidelines.

The team found that the panel had used a similarly inappropriate basis to justify removing the core spray (CS) system from the risk-significant category. The CS system met the NUMARC criteria for risk significance by appearing in the top 90% CDF cutset contribution category. The expert panel determined that since the system "barely" met the criteria and other sources of injection were available (e.g., LPCI, HPSW) the system was not risk significant.

The team found the expert panel process to be weak in that these determinations did not appear to provide supportable bases for deviation from NUMARC and generally accepted probabilistic analysis methods. Additionally, the expert panel's activities were not proceduralized, and documentation was insufficient to reconstruct the basis for the panel's decisions.

Some information obtained during the interviews of the expert panel conflicted with PECO Nuclear's own guidance, i.e., "PECO Energy Approach to Risk Significance." For example, the instrument air system was modeled as a "black box" with a failure probability of 1E-4 based on generic industry data. The actual reliability of the instrument air system at Peach Bottom was not used. Using the NUMARC importance criteria, the instrument air system was risk significant. The expert panel determined that the system was non-risk significant, because other sources of air were available and these were not modeled in the PRA. Some panel members believed that the PRA's failure probability for instrument air was derived from performance of Peach Bottom's instrument air compressors during the 1989-91 time period. During that period, the reliability of the compressors was low, but PECO Nuclear installed new compressors in 1991. The panel members expected the new compressors would lower the failure probability of the instrument air system, and if the improved reliability was used in the PRA, the system would be of low risk significance. The expert appeared to misunderstand the source of data used in the PRA (generic verses plant specific).

c.1 Conclusions on Expert Panel

The team concluded that the expert panel process was weak. The panel did not have a well structured or reproducible process.

Several of the final determinations made by the panel appeared to have an inadequate technical basis. The decisions reached by the panel were not supported by the technical arguments presented to the team. The consequences of these potential misclassifications were that certain SSCs may have been improperly dispositioned from the risk-significant to the nonrisk-significant category. The issue of whether the panel's determinations of risk significance was adequate for the condensate, feedwater, and core spray systems is considered an unresolved item (URI 50-277;278/96-07-02) and will be further reviewed upon receipt of additional information justifying the decision to categorize the systems as non-risk significant.

b.2 Observations and Findings on Risk Ranking

A plant specific PRA was used to rank SSCs with regard to risk significance. NUMARC 93-01 recommendations for RRW, RAW, and CDF cutset contribution were used. To classify SSCs as either risk or non-risk significant, PECO Nuclear used the highest ranked component as a surrogate for the importance of the entire system. PECO Nuclear believed that the NUMARC recommended criteria for risk significance was to be applied to components and not to importance measures of system safety functions. The team noted that PRA models vary in degree of modeling detail and that greatly decomposed (very detailed) models can result in lower relative importance of the basic model elements than if coarser (system or super-component) modeling were used. Therefore a greatly decomposed system might not be determined to be important if the most important component is used as a surrogate for system importance. This is considered a generic issue involving the proper interpretation of NUMARC 93-01. It will remain an open item (IFI 50-277;278/96-07 03) pending the establishment of an NRC position on the interpretation.

The PRA for ranking SSCs predominantly used generic failure probability data. This could result in a less accurate ranking than if plant specific data were used. PECO Nuclear was not collecting reliability data for all risk-significant SSCs and had no plans to update the analysis using plant specific data. PECO Nuclear indicated that the appropriateness of the data had not been checked since 1991. PECO Nuclear collects unavailability data on risk-significant SSCs and uses this information to estimate the impact of SSC unavailability on CDF. However, the potential impact of SSC unavailability on the risk significance ranking was not routinely determined.

PECO Nuclear used a cut set-matching approach to quantifying its PRA model to estimate core damage frequency (CDF). In this approach, individual system fault trees are first solved and then the minimal cut sets from these trees are combined using sequence logic from the event trees. This is a commonly used approximation to a fully linked approach. In solving the fault trees, a truncation level of 1E-10 was used for most fault trees. There were some exceptions imposed by limitations in the software. In combining together fault tree cut sets to obtain sequence cut sets, a truncation level of 1E-11 was used. The overall core damage frequency was approximately 6E-6 per year. The team found this combination of truncation levels to be adequate.

As noted before, PECO Nuclear had used the core damage frequency contribution method of NUMARC 93-01 as one of the risk importance measures. However, instead of using the top 90% of cut sets contributing to CDF, the top 85% was used. As a result of the team's concerns in this area, PECO Nuclear confirmed that no additional SSCs were present when 90% was used as the criterion. Nevertheless, the team considered that, unless PECO Nuclear extends the analysis to 90%, they cannot be sure of identifying all potentially risk-significant SSCs in accordance with the NUMARC guideline. This is because subsequent changes in the reliability data and/or modeling could result in changes in the risk profile.

c.2 Conclusions on Risk Ranking

The team concluded that the use of generic failure probability data was a weakness in the program. The use of the highest risk-significant component as a surrogate for system importance is a generic issue and has been assigned an open item. The use of the top 85% of the cut set contributing to CDF as opposed to the NUMARC guideline of 90% made no difference in this case, but the basis for varying from the guideline was not documented. The approach of using the top 85% of the cut sets contributing to CDF was considered an inappropriate deviation from the NUMARC guidance. The team noted that there was a general problem with lack of documenting decisions and related bases or justification when not following the NUMARC 93-01 guidance. The team identified an open item to address the general concern with inappropriate and poorly documented deviations from NUMARC guidance (IFI 50-277;278/96-07-04).

M1.3 (a)(3) Periodic Evaluations

a. Inspection Scope

Section (a)(3) of the rule requires that performance and condition monitoring activities and associated goals and preventive maintenance activities be evaluated taking into account, where practical, industry-wide operating experience. This evaluation is required to be performed at least one time during each refueling cycle, not to exceed 24 months between evaluations. The team reviewed the procedural guidelines for these evaluations and the first periodic assessment for Unit 3 for the period October 1993 to October 1995. The team also reviewed two quarterly evaluation reports and discussed the activities with the maintenance rule coordinator.

b. Observations and Findings

At the time of this inspection, PECO Nuclear was not required by the rule to have performed the first periodic evaluation. However, PECO Nuclear had established plans and procedures for these evaluations and had performed two quarterly and one biennial evaluation. The team reviewed these reports, which, except as noted, appeared to meet the requirements of the rule.

The team found the biennial report lacked sufficient detail in certain areas, such as:

review of the effectiveness of corrective actions taken to address MPFFs

application of industry-wide operating experience

evaluating the effectiveness of balancing reliability and unavailability

adjustments made to preventive maintenance

summary of SSCs moved from (a)(1) to (a)(2) or vice versa.

It was noted that, besides the maintenance rule-associated quarterly reports, the system managers also trended their system performance with other programs. For example, the preventive maintenance program for instrument nitrogen was adjusted by tracking compressor run time. PECO Nuclear plans to evaluate each unit such that one unit's periodic evaluation would be performed each year.

c. Conclusions

The plans and procedures for performing the periodic evaluations appeared to meet the requirements of the rule. However, the periodic evaluation for Unit 3 lacked details in some important areas as noted.

M1.4 (a)(3) Balancing Reliability and Unavailability

a. Inspection Scope

Paragraph (a)(3) of the rule requires that adjustments be made where necessary to assure that the objective of preventing failures through the performance of preventive maintenance is appropriately balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance. The team reviewed the plans and discussed these plans with the maintenance rule coordinator who is responsible for preparing this evaluation.

b. Observations and Findings

Their approach for optimizing availability and reliability was to assess whether the unavailability performance criteria (in percent) was met. If it was met, then the SSC satisfied management's expectation from an availability standpoint. The team noted that the approach would provide useful information from the perspective of controlling unplanned unavailability at the plant level. PECO Nuclear also assessed the balance between planned voluntary on-line maintenance and unplanned maintenance, since unplanned work was generally an indicator of unavailability. PECO Nuclear used unplanned unavailability as a surrogate for reliability, assuming that higher unplanned unavailability equalled lower reliability. Therefore, instead of balancing reliability and unavailability, they were balancing planned and unplanned unavailability. The team found this approach to be deficient, because information limited to equipment out of service time for corrective maintenance does not give adequate information about system reliability. This was because it does not take into account the number of failures, the number of demands, or the total time in service. Meaningful estimates of reliability would necessitate information that incorporated demands and time in service.

c. Conclusions

The team concluded that the approach to balancing reliability and unavailability would not accomplish the objective of preventing failures of SSCs while also minimizing unavailability as required by the rule. The acceptability of this approach appeared to be doubtful and will be reviewed as an open item. (IFI 50-277;27 :/96-07-05).

M1.5 (a)(3) Plant Safety Assessments Before Taking Equipment Out of Service

a. Inspection Scope

Paragraph (a)(3) of the rule states that the total impact on plant safety should be taken into account before taking equipment out of service for monitoring or preventive maintenance. The team reviewed the applicable procedures and discussed the process and procedures with appropriate PECO Nuclear personnel, including outage planners and licensed operators.

b. Observations and Findings

The program for assessing plant safety during equipment outages is described in Procedure AG-43, "Guideline for the Performance of System Outages." The process provides several mechanisms for assessing and limiting the plant impact of equipment removed from service including: procedural restrictions, which prohibit the concurrent removal of specific systems from service based on their contribution to the probabilistic core damage frequency, technical specification limitations, consultation with the probabilistic safety assessment (PSA) group on an "as needed" basis, multi-disciplined schedule reviews and operating experience. Procedure AG-43 did not specifically address the systems covered by the maintenance rule; those systems were addressed in a separate procedure.

The team found that equipment outage planning personnel and licensed operators were knowledgeable about the AG-43 procedural requirements but were not familiar with the specific systems contained within the scope of the maintenance rule. The team noted that a potential existed for a system to be removed from service without a proper plant safety assessment since:

AG-43 did not address all systems covered by the maintenance rule that perform a plant safety function, and

The outage planning personnel and licensed operators may not identify the need for a PSA review due to their unfamiliarity with the specific systems covered by the rule.

c. Conclusions

Knowledge of the rule by operators and planners was adequate for them to carry out their responsibilities prior to removing equipment from service. Execution of the maintenance rule by operators and planners could be enhanced by improving their awareness of the specific systems covered by the rule. This could provide additional assurance that the plant impact of removing systems from service is consistently and properly assessed.

M1.6 (a)(1) Goal Setting and Monitoring and (a)(2) Preventive Maintenance

a. Inspection Scope

The team reviewed program documents in order to evaluate the process established to set goals and monitor under (a)(1) and to verify that preventive maintenance was effective under (a)(2) of the rule. The team also discussed the program with appropriate plant personnel. The team reviewed the following systems:

(a)(1) Systems

- MO-3-10-025A, low pressure coolant injection (LPCI) valve
- standby gas treatment system (SGTS) for Unit 3
- 4 kV Breakers

(a)(2) Systems

- residual heat removal (RHR)
- emergency service water (ESW)
- instrument nitrogen
- safety grade instrument gas (SGIG)
- feedwater (FW)
- high pressure coolant injection (HPCI)
- reactor core isolation cooling (RCIC)
- electrohydraulic control (EHC)
- automatic depressurization system (ADS)

- main steam SRVs (MSRV)
- reactor recirculation
- primary containment isolation (PCIS)
- emergency diesel generators (EDG)
- 125/250 Vdc batteries
- structures

The team reviewed each of these systems to verify that goals or performance criteria were established in accordance with safety, that industry-wide operating experience was taken into consideration where practical, that appropriate monitoring and trending was being performed, and that corrective actions were taken when an SSC failed to meet its goal or performance criteria or experienced a maintenance preventible functional failure (MPFF). The team also reviewed performance criteria for SSCs not listed above.

b.1 <u>Observations and Findings for Safety Considerations in Setting Goals and Performance Criteria</u>

The maintenance rule as implemented in the NUMARC guidance requires that safety (risk) be taken into consideration when establishing goals under (a)(1) or performance criteria under (a)(2).

As discussed previously in this report, PECO Nuclear used a risk determination process to assess the relative risk of all SSCs within the scope of the rule. The results of this process were used to categorize SSCs as either risk significant or non-risk significant. Performance criteria were established based on these categorizations; however, PECO Nuclear did not always follow the NUMARC guidance.

PECO Nuclear made a distinction between performance indicators and performance criteria. Performance indicators were the parameters used to measure performance, whereas performance criteria were the quantitative standards against which actual performance was measured. In this report, the team did not make this distinction.

In accordance with the NUMARC guidance, system or train level performance criteria are established for all risk-significant SSCs and those nonrisk-significant SSCs that are used in standby service. For risk-significant SSCs, the performance criteria should include both unavailability and reliability. Plant level performance criteria are established for all remaining nonrisk-significant, normally-operating SSCs. The plant level performance criteria includes the following:

- unplanned automatic reactor scrams per 7000 hours critical
- unplanned safety system actuations and/or
- unplanned capability loss factor

b.1.1 Observations and Findings for (a)(1) Goals

Three SSCs had been placed into the (a)(1) category. The team's review of these SSCs is discussed below.

When establishing the SGTS limiting condition for operation (LCO) performance criterion, PECO Nuclear did not account for electrical bus outages, which affect SGTS unavailability. As a result, the LCO performance criterion for Unit 3 was exceeded and the system was placed into (a)(1). PECO Nuclear recognized that unit specific LCO performance criteria did not adequately reflect system operation. The SGTS was divided into filter trains and fan groups to reflect system performance. Using the new LCO performance criteria for the trains and groups did not cause the system to exceed its performance criteria and the system was adequately removed from (a)(1). The team found the approach and logic to be reasonable.

MO-3-10-025A, a LPCI valve was placed into (a)(1) due to repetitive failures. Since the failures were associated with one valve, PECO Nuclear decided to place the component versus the RHR system into (a)(1). PECO Nuclear performed a root cause determination and initiated corrective actions to ensure the valve would function as designed. Goals were established commensurate with a problem motor-operated valve, such as successful completion of static and dynamic VOTES testing. After several successful test results, the valve developed a packing leak. The packing was removed and it was determined the valve stem was scratched, which may have caused the packing leak to develop. The valve was repacked with a different type of packing material and placed back in service. The valve remained in (a)(1) and supplemental goals were established based on the packing leak. These goals included successful completion of quarterly stroke testing with no packing leakage. Other corrective action included revising the stroke testing procedure to equalize pressure on both sides of the valve disc to prevent operating the valve with possible reactor coolant pressure against one side of the valve, which the valve was not designed to open against. The valve was scheduled to be disassembled during the next outage to determine the root cause of the packing leak. The team found the goals and corrective actions implemented for this valve were appropriate.

The 4 kV breakers were classified as an (a)(1) component because of a generic problem associated with grease hardening. The component will remain in (a)(1) until it has been demonstrated that the problem was been solved. The team conducted a review of the breakers and interviewed the system manager. The team found the goals to be acceptable and appropriate.

c.1.1 Conclusions for (a)(1) Goals

The team concluded that the goals established for the (a)(1) systems were acceptable and appropriate.

b.1.2 Observations and Findings for (a)(2) Performance Criteria

MPFFs are being used as a measure of reliability for performance criteria. System managers established the performance criteria using historical data without accounting for the number of demands over a 24-month time period. There was no connection to the reliability numbers used in the PRA. Most SSC performance criteria were two MPFFs per 24 months, which was based on not having one MPFF put the SSC into the (a)(1) category. The core spray performance criteria were three MPFFs per 24 months, because the system had already had one MPFF when the performance criteria was established. The team observed that reliability performance criteria using MPFFs may not be a good measure of reliability since it does not account for system demands. Also, there did not appear to be any effort to relate actual performance history to the reliability numbers used in the PRA. NUMARC guidance indicates that performance criteria for risk-significant SSCs be established to assure that reliability and availability assumptions used in the PRA, or other risk determining analysis are maintained or adjusted when determined necessary by the utility.

A plant level performance criteria of 10 MPFFs per 24 months was established for 17 SSCs on each unit. This could potentially allow an SSC to have nine MPFFs without evaluation as to its inclusion in (a)(1). When this concern was brought to PECO Nuclear's attention, they initiated a review to determine if SSC level performance criteria should also be established.

The unavailability performance criteria were generally close to, although slightly greater than, the unavailability assumptions used in the PRA. PECO Nuclear used the PRA to calculate the CDF increase if risk-significant equipment was assumed to be unavailable at their unavailability performance criteria. This analysis showed only a slight increase in CDF, which was not considered significant by the team.

PECO Nuclear had identified, and the team concurred, that the performance criteria for the PCIS was inadequate. PECO Nuclear had submitted an action request (AR 1039453) to establish a more useful performance criteria for the PCIS. The performance criteria used the percentage of time that the PCIS was operated in a limiting condition of operation (LCO), as stated in the technical specifications. The hours in the LCO were to be totaled and expressed as a percentage of the total hours available and monitored as a rolling average. The PCIS had zero hours operated in an LCO; and, therefore, the performance criteria provided no actual data to determine an indication of system performance.

Based upon a review of specific SSC performance criteria the team found the following concerns:

- Safety Grade Instrument Gas (SGIG), the CRD standby pump, RPS, and the EDG building ventilation are risk-significant standby systems, which had performance criteria for train level MPFFs established, but none for unavailability.

- Control Room Emergency Ventilation (CREV) is a risk-significant standby system, which had no performance criteria established.

- Reactor Recirculation, EHC, and Feedwater are nonrisk-significant systems with performance criteria established at the plant level. However, the performance criteria did not include unplanned capability loss factor, which was needed to measure functional performance.

  These systems could experience repeated maintenance problems that could only be effectively monitored at the plant level by the unplanned capability loss factor. The unplanned capability loss factor was not used for any system monitored at the plant level. The team noted that three Unit 3 reactor recirculation system pump trips had occurred over approximately the last two years that were not considered functional failures since the pump trips did not result in a reactor shutdown. After discussion with the team, PECO Nuclear indicated that unplanned capability loss factor would be incorporated into performance criteria monitoring for the reactor recirculation, EHC, feedwater, condensate and circulating water systems.

- Adequate performance criteria for the EHC, turbine bypass valves, and MSRV systems were not implemented to monitor the maintenance rule function (reactor pressure control) of these systems. These systems were only monitored using the plant level criteria of unplanned shutdowns.

- The MSRV system is a risk-significant standby system, which was only monitored at the plant level using unplanned shutdowns.

c.1.2 <u>Conclusions for (a)(2) Performance Criteria</u>

The team concluded that PECO Nuclear deviated from the NUMARC guidance, in a number of cases, with no supporting justification to show that the approach was equivalent.

The team found that a number of performance criteria were inadequate to monitor performance in accordance with the maintenance rule. SSCs identified with inadequate performance criteria included:

safety grade instrument gas
standby CRD pump
reactor protection system
emergency diesel generator building ventilation
control room emergency ventilation
reactor recirculation

electrohydraulic control system
feedwater
turbine bypass valves
main steam SRVs

10 CFR 50.65(a)(1)and (a)(2) require that the performance or condition of structures, systems, and components shall be monitored against licensee-established goals unless "it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance..." However, PECO Nuclear was not monitoring the performance or condition of numerous systems and components against established goals, nor had PECO Nuclear demonstrated the effectiveness of preventive maintenance on these systems and components. The monitoring of the effectiveness of preventive maintenance had not been demonstrated in that the expectation is that system performance criteria will address both reliability and availability. However, PECO Nuclear did not account for system demands as part of reliability criteria, did not have availability criteria for some systems, did not have availability or reliability criteria for the control room emergency ventilation system, did not use the appropriate criteria of unplanned capability loss factor for some systems, did not have criteria for all maintenance rule functions for some systems, and did not monitor MSRVs at the appropriate system level. Accordingly, these examples represent an apparent violation of 10 CFR 50.65 (EEI 50-277;278/96-07-06).

Further, Regulatory Guide 1.160 endorsed NUMARC 93-01 as an acceptable method for complying with 10 CFR 50.65. PECO Nuclear established their maintenance rule program in accordance with NUMARC 93-01, but did not provide justification for deviations from NUMARC 93-01. In several examples noted above, performance criteria, as implemented, conflicted with NUMARC 93-01.

b.2 <u>Observations and Findings for Use of Industry-Wide Operating Experience</u>

The maintenance rule, as implemented using the guidance in NUMARC 93-01, specifies that industry-wide operating experience be taken into consideration, where practical, when establishing goals or performance criteria.

Based upon reviews of documentation and discussions with PECO Nuclear personnel, the team determined that PECO Nuclear had established programs for reviewing and evaluating operational experience. Procedure LR-C-04, "Operating Experience Assessment Program," assigned responsibility for review and tracking of industry experience. Procedure AG-CG--28.1, "Maintenance Rule Implementation Program," required the system managers to consider maintenance rule requirements during their review of industry operating experience. The system managers who have responsibility for establishing performance criteria for their systems reviewed industry experience to assess the effect on their systems and to implement corrective actions as appropriate. The team found that system managers were able to identify system improvements implemented from use of industry operating experience, but no specific examples were identified where this experience influenced goals or performance criteria. The team noted that several feedwater system modifications were planned or under consideration as a result of industry experience.

c.2 Conclusions for Use of Industry-Wide Operating Experience

Industry-wide operating experience had been incorporated into the maintenance program. Nonetheless, evidence was lacking that industry experience was or would be taken into account, where practical, when setting goals or performance criteria. This represents an open item. (IFI 50-277; 278/96-07-07)

b.3 Observations and Findings for Monitoring and Trending

NUMARC guidance indicates that monitoring will be performed to determine if maintenance of (a)(1) SSCs results in acceptable performance. For (a)(2) SSCs performance is trended against the established performance criteria so that adverse trends can be identified. The objective of monitoring plant level performance criteria is to focus attention on the aggregate performance of many of the operating SSCs that are not individually risk significant.

The Peach Bottom system managers are responsible for trending and evaluating SSC performance trended at the train or system level, and the maintenance rule coordinator is responsible for SSCs trended at the plant level.

The team reviewed the quarterly and periodic evaluation reports that trended performance. The quarterly report appeared to be a good method to disseminate SSC performance information to plant management. It was noted by the team that goals for (a)(1) systems were not explicitly identified in the reports. These goals were documented in plant enhancement program documents (PEPs). This could indicate that the maintenance rule program was not fully integrated into overall maintenance activities.

Several system managers had different understanding of the performance criteria as related to evaluating whether the SSC should be placed into (a)(1). For example, if the performance criteria were two MPFFs per 24 months, some system managers stated that when two MPFFs were reached the system was considered for (a)(1), whereas other system managers believed that three MPFFs were required before considering (a)(1). PECO Nuclear's position was subsequently clarified to consider (a)(1) when the performance criteria are exceeded at three MPFFs. Some confusion may have also arisen over the inconsistent manner that the performance criteria were stated. For example, RHR performance criteria were stated as two MPFFs per 24 months, while SGIG was given as <two MPFFs per 24 months. Also, some system managers were not sure how the unavailability performance criteria would be applied in assessing SSC performance.

During the team's review of the RHR system, an apparent MPFF was identified by the team that was not previously identified. The MPFF involved a tagout to replace relays in the reactor water cleanup system that caused a loss of shutdown cooling. The RHR shutdown cooling function was lost and should have been maintenance preventable if the tagout was correct. A review of performance of the RHR shutdown cooling system found that the performance criteria were not exceeded with this MPFF counted. PECO Nuclear was planning to further review this issue.

The team noted that system managers were using predictive monitoring and trending to assess performance and determine when preventive maintenance was required. For example, the instrument nitrogen system manager scheduled compressor oil changes and inspections by trending compressor run time.

c.3 Conclusions for Monitoring and Trending

The team concluded that the problems with adequate performance criteria and the use of these criteria made it difficult to adequately monitor and trend performance. System managers had an inconsistent understanding of what was needed to exceed a performance criterion.

b.4 Observations and Findings for Corrective Actions

The team reviewed procedures for establishing corrective actions and reviewed the corrective actions taken for a sample of the systems listed in Section M1.6.a of this report. The team interviewed each system manager who had responsibility for establishing corrective actions. The corrective actions were considered by the team to be effective. Functional failures were properly classified as MPFFs. The team reviewed the performance enhancement program (PEP) issue 10002183, which discussed an unexpected Unit 2 reactor recirculation pump speed increase that resulted in a reactor scram. In addition to performing troubleshooting and failure analysis for the failed component, PECO Nuclear identified several generic related causal factors. Corrective actions included replacement of the defective hardware, installation of a temporary modification to improve system monitoring, and installation of pump mechanical speed limiting stops on the opposite unit.

The team noted that a number of feedwater system problems had occurred over the past approximately two year period. Unit 2 experienced five recirculation system runbacks and two reactor water level perturbations due to the feedwater system.

Unit 3 experienced one reactor scram and nine reactor water level transients or perturbations due to the feedwater system. Extensive feedwater governor control system modifications were planned to improve system reliability.

c.4 Conclusions for Corrective Actions

The team concluded that in general corrective actions were appropriate; however, their effectiveness could not be determined in some cases until SSC changes were implemented.

## M2 Maintenance and Material Condition of Facilities and Equipment

a. Inspection Scope

In the course of verifying the implementation of the maintenance rule using IP 62706, the team performed walkdowns to examine the material condition of the following systems:

Structures (limited)
Reactor Recirculation
Electrohydraulic Control
High Pressure Coolant Injection
Reactor Core Isolation Cooling
Feedwater
Residual Heat Removal
Emergency Service Water

b. Observations and Findings

Except as noted, the systems were free of corrosion, oil leaks, water leaks and trash, and based upon external condition, appeared to be well maintained. The HPCI and RCIC systems appeared to be in very good condition. Minor oil leaks were noted on the Unit 3 feedwater pumps. The Unit 2 feedwater system had a larger number of leaks. The team noted, however, that Unit 2 was nearing its next planned outage and Unit 3 had only recently come out of an outage.

c. Conclusions

In general, the material condition of the systems examined was excellent.

## M4 Staff Knowledge and Performance

### M4.1 Knowledge of the Maintenance Rule

a. Inspection Scope (62706)

The team interviewed engineers and engineering managers to assess their understanding of the maintenance rule and associated responsibilities. Also, the team interviewed licensed reactor and senior reactor operators to determine if they understood the general requirements of the rule and their particular duties and responsibilities for its implementation.

b. Observations and Findings

The system managers were very knowledgeable of their systems and were familiar with related industry operating experience. System managers and other engineers and managers were familiar with the maintenance rule requirements. However, in some cases they were not sure how to apply the SSC performance criteria, as discussed in b.3. Also, in one case, the system manager supervisor did not appear to be cognizant of system performance for systems under his control.

The team interviewed four licensed operators and noted that they were familiar with the maintenance rule and their role in its implementation. The operators indicated that their primary duties included review of maintenance plans and schedules, and timely removal and restoration of equipment to maximize its availability. The team found the operators were not familiar with which systems were within the scope of the maintenance rule, as noted in Section M1.5.

c. Conclusions

All engineers, engineering managers, and licensed operators were knowledgeable of their assigned systems and demonstrated sufficient knowledge to adequately implement their responsibilities under the maintenance rule. However, some weaknesses in knowledge of certain aspects of implementing the rule were noted.

**M7 Quality Assurance (QA) in Maintenance Activities**

Self-Assessments of the Maintenance Rule Program

The team reviewed the following assessments:

(1) Self-Assessment by Nuclear Engineering Division (NED) Engineering Assurance Branch, conducted 6/24-26/96.

(2) QA Surveillance Report, PSR-95-282, conducted 11/20-22/95 and 12/4-15/95.

(3) Independent Safety Engineering Group (ISEG) Assessment of Functional Failures (FFs) and MPFFs for Limerick, dated 10/27/95.

(4) MRITE Assessment of the Maintenance Rule Program at Peach Bottom, conducted 3/7-9/95.

The assessments identified both good performance areas and areas in need of management attention. Several areas noted by the assessments as needing attention were also identified as concerns by the team, including: (1) risk ranking by the expert panel, (2) weak documentation, (3) procedure adequacy, and (4) performance criteria for the PCIS system. Several examples were noted where assessment findings had been acted upon, such as the performance criteria for the PCIS system and the scope of structures. PECO Nuclear indicated that for the most recent assessment they had not had time to take appropriate actions.

The team concluded that the assessments provided meaningful feedback to management. This feedback had resulted in program improvements, which are still ongoing. The team noted that some problems with the maintenance rule program were not identified in these self-assessments.

## V. Management Meetings

### X1 Exit Meeting Summary

The team discussed the progress of the inspection with PECO Nuclear representatives on a daily basis and presented the inspection results to members of management at the conclusion of the inspection on August 9, 1996. PECO Nuclear acknowledged the findings presented.

The team asked PECO Nuclear whether any materials examined during the inspection should be considered proprietary. PECO Nuclear indicated that the four program assessments provided to the team were proprietary information.

## Partial List of Persons Contacted

### PECO NUCLEAR

T. Mitchell, VP-Peach Bottom Atomic Power Station
G. Edwards, Plant Manager
J. McElwain, Director, Outage Management
C. Swenson, Senior Manager Outages
J. Armstrong, Senior Manager Plant Engineering
O. Limpias, Manager Civil/Structural Design Branch
T. Wasong, Manager Operations Services
J. Hufnagel, Manager Performance and Reliability
A. Marie, Manager PSA Branch
L. Cobosco, Maintenance Rule Coordinator
G. Krueger, PSA Engineer
J. Jordan, Training Manager
R. Smith, Inspection Coordinator

### NRC TEAM SUPPORT

J. Wilcox
D. Taylor
F. Talbot
D. Coe

## List of Inspection Procedures Used

IP 62706 Maintenance Rule

List of Items Opened, Closed, and Discussed

50-277;278/96-07-01 (URI) Identification of the alternate ac power source components to be included within the scope of the rule.

50-277;278/96-07-02 (URI) The adequacy of the basis for the expert panel's determination of risk significance for the condensate, feedwater, and core spray systems.

50-277;278/96-07-03 (IFI) Interpretation of NUMARC 93-01 to allow use of components verses systems for risk ranking and determining risk significance of SSCs.

50-277;278/96-07-04 (IFI) Concern regarding inappropriate and poorly documented deviations from NUMARC 93-01.

50-277;278/96-07-05 (IFI) Adequacy of approach to balancing reliability and unavailability in the periodic evaluation.

50-277;278/96-07-06 (EEI) Adequate performance criteria were not established for a number of SSCs to allow monitoring performance in accordance with the maintenance rule.

50-277;278/96-07-07 (IFI) No evidence existed that industry operating experience had been used in setting goals and performance criteria.

## List of Acronyms Used

| | |
|---|---|
| ADS | Automatic Depressurization System |
| AR | Action Request |
| BOP | Balance of Plant |
| CDF | Core Damage Frequency |
| CFR | Code of Federal Regulations |
| CRD | Control Rod Drive |
| CREV | Control Room Emergency Ventilation |
| CS | Core Spray |
| EDG | Emergency Diesel Generators |
| EHC | Electrohydraulic Control |
| ESW | Emergency Service Water |
| FF | Functional Failure |
| FW | Feedwater |
| HPCI | High Pressure Coolant Injection |
| HPSW | High Pressure Service Water |
| IFI | Inspection Followup Item |
| IPE | Individual Plant Evaluation |
| ISEG | Independent Safety Engineering Group |
| kV | Kilovolts |
| LCO | Limiting Condition for Operation |
| LPCI | Low Pressure Coolant Injection |
| MPFF | Maintenance Preventable Functional Failure |
| MS | Main Steam |
| MSRV | Main Safety Relief Valve |
| NOV | Notice of Violation |
| NRR | Nuclear Reactor Regulation |
| PCIS | Primary Containment Isolation System |
| PEP | Plant Enhancement Program |
| PRA | Probabilistic Risk Assessment |
| PSA | Probabilistic Safety Assessment |
| QA | Quality Assurance |
| RAW | Risk Achievement Worth |
| RCIC | Reactor Core Isolation Cooling |
| RHR | Residual Heat Removal |
| RPS | Reactor Protection System |
| RRW | Risk Reduction Worth |
| SGIG | Safety Grade Instrument Gas |
| SGTS | Standby Gas Treatment System |
| SRV | Safety Relief Valve |
| SSC | Structures, Systems or Components |
| URI | Unresolved Item |